Steve

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated April 7, 2009, as supplemented by letters dated September 11 and October 9, 2009 (Agencywide Documents Access and Management System Accession Nos. ML091250564, ML092570205, and ML092860098), FPL Energy Point Beach, LLC, submitted a request to increase each unit's licensed core power level from 1540 megawatts thermal (MWt) to 1800 MWt reactor core power, and revise the technical specifications to support operation at this increased core thermal power level.

The Reactor System Branch has reviewed the information provided and determined that in order to complete its evaluation, additional information is required. We would like to discuss the questions, in draft form below, with you in a conference call.

This e-mail aims solely to prepare you and others for the proposed conference call. It does not convey a formal NRC staff position, and it does not formally request for additional information.

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2.8.5.0-1.   Explain why a +1.4°F RCS T_{avg} bias was applied for RTDP analyses rather than treated statistically. Explain the effect of this assumption and compare to a statistical treatment of the same conditions.

2.8.5.0-2.   Please provide a copy of Reference 6 (page 2.8.5.0-13), Nuclear Safety Advisory Letter NSAL-07-10, Loss-of-Normal Feedwater/Loss-of-Offsite AC Power Analysis PORV Modeling Assumptions, November 7, 2007

2.8.5.1-1.   How were the limiting break sizes, 0.59 ft2 (Unit 1) and 0.63 ft2 (Unit 2), determined for the analyses of Steam System Piping Failures at Full-Power?

2.8.5.1-2.   Describe the reactor vessel inlet mixing assumptions, and their bases, used in the analyses of Steam System Piping Failures and other asymmetric cooldown events.

2.8.5.1-3.   Since the OPΔT trip function is not qualified for a harsh environment caused by the steam line break, the applicant states that the Hi-1 containment pressure safety injection signal would generate a reactor trip signal before the time credited, in the

analyses, for the OPΔT trip signal. What is the basis for this statement? What models and assumptions were used in containment pressure response analyses in order to yield conservatively late Hi-1 containment pressure safety injection signals.

2.8.5.1-4.   What are the results of analyses for inside containment cases of Steam System Piping Failures at Full-Power that credit only the low steam line pressure safety injection signal?

2.8.5.1-5.   During a full power-steam line-rupture-core-response event, the main feedwater system flow will increase to match the steam flow until feedwater isolation occurs. It appears that the main feedwater system flow does not increase to match the steam flow, during a no-load steam line-rupture-core-response event. Describe how such a feedwater system response would affect a 1.4 ft2 no-load steam line break.

2.8.5.1-6.   The steamline break analysis discussion states that the core attains criticality before boron solution from the ECCS and accumulators enters the RCS (actually the core). Tables 2.8.5.1.2-1 and 2.8.5.1.2-2 indicate that the core attains criticality sometime after flow from the ECCS enters the RCS and shortly after the accumulators begin to inject. This indicates that the delivery of boron solution is determined by RCS pressure (not ECCS actuation signal delays or pump startup and valve opening times), and comes largely from the accumulators. (Question continues on next page)

2.8.5.1-6   (Continued)

    a) When does boron solution from the safety injection system enter the core?
    b) What would be the result of a smaller steamline break, that would not depressurize the RCS to the accumulator injection setpoint?
    c) What would be the result of an even smaller steamline break that would not depressurize the RCS, to the safety injection system shutoff head?

2.8.5.2-1.   In the Loss of Load event discussion, one of the listed acceptance criteria is stated as follows, "An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, fuel failure is assumed for all rods for which the DNBR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model, that fewer failures occur. There is no loss of function of any fission product barrier other than the fuel cladding." Why are there no analyses of AOO combinations presented to show this criterion has been satisfied?

2.8.5.2-2.   Please explain the assumptions regarding AFW flow: The AFW flow was initiated 30 seconds after the low-low SG water level setpoint was reached; from 30 to 60 seconds, the AFW flowrate ramped from 0% to 80% of total flow; from 60 to 120 seconds, the AFW flowrate ramped from 80% to 100% of total flow; beyond 120 seconds, 100% of total flow (275 gpm) was maintained.

2.8.5.2-3.   If the restrictive acceptance criterion, that the pressurizer does not become water solid, were used for the Loss of Feedwater event, then why were the PORVs not modeled?

2.8.5.4.5-1.   Please explain how, and in which operating modes, the Chemical and Volume Control System (CVCS) is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

2.8.5.6-1.   Explain whether transient local and core-wide oxidation values calculated for the large and small break LOCA analyses include pre-transient oxidation. Discuss whether, when considering pre-transient oxidation, results remain within the 50.46 acceptance criteria.

2.8.5.6-2.   Discuss whether oxidation models for ASTRUM and NOTRUMP calculate cladding oxidation on both inner and outer cladding surfaces.

2.8.5.6-3.   Provide the result for core-wide oxidation for the SBLOCA analysis.

2.8.5.6-4.   The AOR SBLOCA analysis predicts a limiting PCT of 1205°F for the Unit 1 3-inch break. The EPU SBLOCA analysis predicts a limiting PCT of 1103°F. Compare assumed initial conditions, equipment functional capabilities and actuation setpoints to identify the causes of this significant decrease.

2.8.5.6-5.   In light of the fact that NRC approval of the SBLOCA analysis approval is required prior to EPU implementation, provide information to justify that any analytic changes

effecting the noted reduction in PCT are the direct result of modifications that will be implemented following interim approval of requested EPU-related modifications.

2.8.5.6-6.   Explain what/how the SBLOCA analysis reflects revisions requested to TS 3.3.2, Function 6.e, TS 3.7.3, TS 3.7.5, and TS 3.7.6. Note that the original description of these change requests provided in the April 7, 2009, application for the EPU did not identify these TS Changes as affecting the SBLOCA analysis.

2.8.5.7-1.   List all operator actions credited for the ATWS analysis.

2.8.5.7-2.   For the period of time 250-300s following the ATWS initiating event, when primary system pressure and temperature are decreasing, explain what phenomena hold down reactivity.

2.8.5.7-3.   Explain what provides long-term shutdown capability following the ATWS.

2.8.5.7-4.   SRP Chapter 15.8 indicates that ATWS analytic results should be compared to the LOCA acceptance criteria provided in 10 CFR 50.46, regarding coolable geometry, peak cladding temperature, cladding oxidation, and hydrogen generation. Demonstrate conformance to these acceptance criteria.

2.8.5.7-5.   Describe how the loss of normal feedwater event is confirmed to be the limiting ATWS transient. Why is no other ATWS initiator evaluated?

2.8.5.7-6.   Explain the reactor coolant system flow transient. Describe the cause of the reduction in flow from 50-140 seconds, and provide the cause of the reactor coolant pump trip that follows the initial flow decrease.

2.8.5.7-7.   The Licensing Report states, "The ATWS evaluation for EPU assumed a PBNP-specific MTC of -8pcm/°F that bounds 95% of the cycle. This value is consistent with that assumed in generic ATWS analyses."

Explain what compensating phenomena and operator actions provide acceptable reductions in risk from ATWS when the MTC is non-bounding of cycle operation.

2.8.5.7-8.   Provide a list of initial conditions and plant parameters assumed in the ATWS analysis that differ from Point Beach-specific parameters. Provide the assumed value and compare it to the actual value that would be appropriate for analysis of Point Beach. Where any parameters are non-bounding, justify their use.

2.8.5.7-9.   Verify that actual component and actuation setpoint testing supports Technical Specification actuation values, and that the Technical Specification actuation values are used in the ATWS analyses. Specifically identify any differences between analyzed equipment actuation setpoints and respective TS values.

2.8.5.7-10.   Confirm whether TS-permitted equipment out of service assumptions are reflected in the ATWS analyses.

2.8.A-1.   During its review of the RAVE implementation for Locked Rotor – Rods in DNB analyses, the NRC staff requested additional information concerning the qualification of RAVE analysts (RAI 1), the nodalization of the VIPRE, RETRAN and SPNOVA models (RAI 2), and sensitivity analyses performed to demonstrate the conservatism of assumed voiding present in excess of 30 percent (RAI 7). Confirm that the information presented in response to the NRC staff's previous request for additional information is applicable to the EPU analyses, as well. If this is not the case, justify any differences from the previous response.

2.8.A-2.   Condition 3 for implementation of RAVE requires that a plant be licensed to use each of the three constituent codes – VIPRE, RETRAN, and SPNOVA – for safety analyses. Detailed justification was provided for EPU implementation of VIPRE and RETRAN; analogous information was not provided for SPNOVA. Please provide the following information:

a) Has SPNOVA been implemented previously at Point Beach for licensing applications? If so, please describe the application and reference its NRC approval.
b) If SPNOVA has not been implemented previously at Point Beach for licensing applications, provide justification for its implementation that is analogous to that provided for implementation of VIPRE and RETRAN.

T.S.-1.   Low pressurizer pressure is credited to terminate the RCCA drop accident. Note 2 of Table 2.8.5.0-5 states, "The generic two-loop RCCA drop analysis, which is applicable to PBNP, modeled the low pressurizer pressure reactor trip setpoint as a "convenience trip." The cases that actuated this function assumed dropped rod and control bank worth combinations that were non-limiting with respect to DNB. The fact that the plant-specific low pressurizer pressure reactor trip setpoint is lower than the value assumed in the generic analysis does not invalidate the applicability of the generic two-loop RCCA drop analysis to PBNP. Therefore, the low pressurizer pressure reactor trip setpoint value that was used in the generic two-loop RCCA drop analysis does not represent an analytical limit for this function for PBNP."

a) Please provide an electronic copy of WCAP-11394-P-A.
b) What trip is credited in the DNB-limiting case?
c) What limiting analysis is available to demonstrate the acceptability of a pressurizer low pressure trip setpoint of 1855 psia?

T.S.-2.   Are ECCS subsystem boron concentration levels proposed to change as a part of the expedited modifications? If not, explain how compliance with 10 CFR 50.36(c)(2)(ii)(B) is maintained in light of post-LOCA subcriticality analyses that presumably credit the increased TS minimum ECCS boron concentrations.

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